

March 16, 1995

Mr. Jerry W. Yelverton  
Vice President, Operations ANO  
Entergy Operations, Inc.  
Route 3 Box 137G  
Russellville, Arkansas 72801

SUBJECT: ISSUANCE OF AMENDMENT NO.178 TO FACILITY OPERATING LICENSE  
NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M90608)

Dear Mr. Yelverton:

The Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 30, 1994, as supplemented by letter dated March 9, 1995.

The amendment relocates refueling cycle specific parameters from the TSs to the core operating limits report as per recommendations promulgated by NRC Generic Letter 88-16. Additionally, the amendment adds a 24 hour limit on operations when only one reactor coolant pump is operating in each loop.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by:  
George Kalman, Senior Project Manager  
Project Directorate IV-1  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No.178 to DPR-51  
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

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Vice President, Operations ANO  
Energy Operations, Inc.  
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A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

A handwritten signature in black ink, appearing to read "George Kalman".

George Kalman, Senior Project Manager  
Project Directorate IV-1  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 178 to DPR-51  
2. Safety Evaluation

cc w/encls: See next page

Mr. Jerry W. Yelverton  
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

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Pope County Courthouse  
Russellville, Arkansas 72801



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

ENERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178  
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Entergy Operations, Inc. (the licensee) dated August 30, 1994, as supplemented by letter dated March 9, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 178, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



George Kalman, Senior Project Manager  
Project Directorate IV-1  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Attachment: Changes to the  
Technical Specifications

Date of Issuance: March 16, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 178

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES

INSERT PAGES

7	7
8	8
9b	9b
11	11
12	12
14b	14b
15	15
16	16
31	31
46	46
47	47
142	142

## 2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS, REACTOR CORE

#### Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical.

#### Objective

To maintain the integrity of the fuel cladding.

#### Specification

- 2.1.1 The maximum local fuel pin centerline temperature shall be  $\leq 5080 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU})^\circ\text{F})$  for TACO2 applications and  $\leq 4642 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})^\circ\text{F})$  for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.
- 2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.
- 2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Safety Limit shown in Figure 2.1-1.

#### Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (3) with potential fuel densification effects:

$$F_q^N = 2.83; \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.65.$$

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by a nuclear power peaking factor of  $F_q^N = 2.83$  or the combination of the radial peak, axial peak and position of the axial peak that yields no less than the DNBR limit.
2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for curves 1, 2, and 3 of Figure 2.1-3 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line the Safety Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2) (1) or 26 percent (BWC) (2).

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## 2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

### Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

### Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

### Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoints for Axial Power Imbalance as given in the COLR.

### Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

### Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

#### A. Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by the value specified in the COLR for a 1 percent flow reduction.

#### B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant

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Table 2.3-1  
Reactor Protection System Trip Setting Limits

	<u>Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)</u>	<u>Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)</u>	<u>One Reactor Coolant Pump Operating in Each Loop<sup>(e)</sup> (Nominal Operating Power, 49%)</u>	<u>Shutdown Bypass</u>
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0 (a)
Nuclear Power based on flow <sup>(b)</sup> and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max <sup>(c)</sup>	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720 (a)
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	13.89 T <sub>out</sub> -6766 (d)	13.89 T <sub>out</sub> -6766 (d)	13.89 T <sub>out</sub> -6766 (d)	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

(a) Automatically set when other segments of the RPS (as specified) are bypassed.

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

(d) T<sub>out</sub> is in degrees Fahrenheit (F).

(e) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

### 3.1 REACTOR COOLANT SYSTEM

#### Applicability

Applies to the operating status of the reactor coolant system.

#### Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

#### 3.1.1 Operational Components

#### Specification

##### 3.1.1.1 Reactor Coolant Pumps

- A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.
- B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

##### 3.1.1.2 Steam Generator

- A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

##### 3.1.1.3 Pressurizer Safety Valves

- A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.
- B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

##### 3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

##### 3.1.1.5 Reactor Coolant Loops

- A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

### 3.1.8 Low Power Physics Testing Restrictions

#### Specification

The following special limitations are placed on low power physics testing.

#### 3.1.8.1 Reactor Protective System Requirements

- A. Below 1720 psig, shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.
- B. Above 1800 psig, nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.8.2 Startup rate rod withdrawal hold (1) shall be in effect at all times.

3.1.8.3 During low power physics testing the minimum reactor coolant temperature for criticality shall be to the right of the criticality limit of Figure 3.1.2-2. The shutdown margin shall be maintained greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn.

#### Bases

The above specification provides additional safety margins during low power physics testing.

#### REFERENCES

- (1) FSAR, Section 7.2.2.1.3.

### 3.5.2 Control Rod Group and Power Distribution Limits

#### Applicability

This specification applies to power distribution and operation of control rods during power operation.

#### Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

#### Specification

- 3.5.2.1 The available shutdown margin shall be greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, immediately initiate and continue boron injection until the required shutdown margin is restored.
- 3.5.2.2 Operation with inoperable rods:
1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.
  2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR. Boron may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.
  3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that an available shutdown margin greater than or equal to that specified in the COLR exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the Hot Standby condition until this margin is established.
  4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.
  5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Power Tilt:

1. Except for physics tests, if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of the tilt limit.
2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following adjustments in setpoints and limits shall be made:
  - a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.
  - b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.
  - c. The reactor power imbalance setpoints shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.
3. If quadrant power tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.
4. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

2.1 Safety Limits, Reactor Core -- Axial Power Imbalance protective limits

2.3.1 Reactor Protection System trip setting limits -- Protection System Maximum Allowable Setpoints for Axial Power Imbalance

3.1.8.3 Minimum Shutdown Margin for Low Power Physics Testing

3.5.2.1 Allowable Shutdown Margin limit during Power Operation

3.5.2.2 Allowable Shutdown Margin limit during Power Operation with inoperable control rods

3.5.2.4 Quadrant power Tilt limit

3.5.2.5 Control Rod and APSR position limits

3.5.2.6 Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated August 30, 1994, as supplemented by letter dated March 9, 1995, Entergy Operations, Inc. (the licensee) proposed changes to the Technical Specifications (TSs) for the Arkansas Nuclear One - Unit 1 (ANO-1). The proposed changes relocate cycle-specific core operating limits from the TSs to the core operating limits report (COLR). The changes were submitted in accordance with the guidance provided in Generic Letter (GL) 88-16. The use of the COLR for ANO-1 was previously approved by the NRC in response to an amendment request dated November 7, 1991. This amendment was issued on April 14, 1992 and transferred several cycle-specific limits from the TSs to the COLR. The current changes are essentially a continuation of the process that began in 1991 and 1992.

In addition to relocating cycle-specific limits, the proposed TS change includes revised conditions when safety limits are applicable and adds a limit of 24 hours for continued operation with only one reactor coolant pump operating in each loop while the reactor is critical. The March 9, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Core Operating Limits Report

The licensee's proposed core operating limit changes are in accordance with the guidance provided by GL 88-16 and are addressed below.

(1) In addition to the cycle-specific core operating limits that were relocated to the COLR by license amendment dated April 14, 1992, the following specifications are being relocated to the COLR as part of this proposed amendment.

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- (a) Specification 2.1, axial power protective limits as defined by Figure 2.1-2
- (b) Specification 2.3.1, 2.5.2.4.2 and Table 2.3-1, protection system setpoints for power imbalance as defined by Figure 2.3-2 and Table 2.3-1
- (c) Specification 3.1.8.3, 3.5.2.1, 3.5.2.2.2, and 3.5.2.2.3, minimum shutdown margin

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR.

(2) The proposed amendment also adds a list of all parameters that were relocated from the TSs to the COLR. This list was added to the administrative controls section of the TSs, Specification 6.12.3.

The NRC staff concludes that the above changes were proposed in accordance with guidance contained in GL 88-16. The changes eliminate cycle-specific limits from the TSs and thus eliminate the need for TS amendments after every refueling. Reactor operations after each refueling continue to be restricted by core operating limits. However, these operating limits are now contained in the COLR. The COLR values of cycle-specific parameters are derived by the licensee using NRC-approved methodologies. The staff finds that this change does not reduce the level of safety. Accordingly, the staff finds that the proposed changes are acceptable.

## 2.2 Reactor Core Safety Limits

The applicability of TS 2.1 is revised from current "during power operation of the plant" to proposed "when the reactor is critical." Since the proposed applicability results in a more restrictive requirement than currently exists, this revision is acceptable. TS 2.1 is also revised to add safety limit requirements for the maximum local fuel pin centerline temperature and the departure from nucleate boiling ratio (DNBR). These revisions change the specifications in a conservative direction and are therefore acceptable.

## 2.3 Reactor Coolant Pumps

TS 3.1.1.1.A has been revised to restrict operation to 24 hours when one reactor coolant pump is operating in each loop and the reactor is critical. This revision is acceptable since it follows recommendations contained in the topical report BAW-10103, Revision 3, and is a more restrictive TS requirement.

### 3.0 CONCLUSION OF TECHNICAL ISSUES

The staff has reviewed the request by Entergy Operations, Inc. to revise the ANO-1 TSs. The proposed revisions relocate cycle specific core parameters from the TSs to the COLR and add more restrictive requirements for operations. The staff concludes that these changes are not adverse to safety and are therefore acceptable.

### 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

### 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 3672). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

### 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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